Clinton Power Station
8401 Power Road
Clinton, IL 61727

Exelon Generation®

U-604407
April 5, 2018

10 CFR 50.73
SRRS 5A.108

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Licensee Event Report 2017-007-02

Enclosed is Licensee Event Report (LER) 2017-007-02: Manual Reactor Scram due to Loss of Feedwater Heating. This is the supplemental report to LER 2017-007-01 dated November 9, 2017. The updated information in the LER is denoted by revision bars located in the right-hand margin. This report is being submitted in accordance with the requirements of 10 CFR 50.73.

There are no regulatory commitments contained in this report.

Should you have any questions concerning this report, please contact Mr. Dale Shelton, Regulatory Assurance Manager, at (217) 937-2800.

Respectfully,

Theodore R. Stoner
Site Vice President
Clinton Power Station

KP/cac

Attachment: License Event Report 2017-007-02

cc:
Regional Administrator – Region III
NRC Senior Resident Inspector — Clinton Power Station
Office of Nuclear Facility Safety — Illinois Emergency Management Agency

IE22
NRR

NRC FORM 366 (04-2017) **U.S. NUCLEAR REGULATORY COMMISSION**

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Clinton Power Station, Unit 1	05000461	1 OF 4

4. TITLE
Manual Reactor SCRAM due to Loss of Feedwater Heating

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
06	10	2017	2017	- 007	- 02	04	05	2018	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 098

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☒ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
	☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Mr. Dale Shelton	(217)-937-2800

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SJ	LA	Moore	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On June 10, 2017, at 2256 CDT, Clinton Power Station (CPS) experienced a complete loss of the 'A' feedwater (FW) heater string. The operators received numerous FW trouble alarms on FW string 'A' and low pressure heater 1A/1B bypass opened (1CB004). The operators entered procedure CPS 4005.01, "Loss of FW Heating," which directs the operators to restore and maintain power at or below the original power level. The operators lowered core flow and inserted all CRAM rods, and then observed that FW temperature had dropped greater than 100°F. As directed by CPS 4005.01, at 2306 hours the reactor mode switch was placed into the shutdown position and Procedure 4100.01, "Reactor Scram," was entered. All systems operated as expected following the scram. At 0100 EDT on June 11, 2017, Event Notification 52800 was made. The loss of FW heating transient was caused by a loss of power to Moore trip units caused by a shorted condition on the Moore trip unit associated with the Hi-Hi level in the 4A FW heater. The root cause is that the design of the FW heater level control trip circuitry was not adequate to prevent scrams due to an unevaluated single point vulnerability. Prior to startup, CPS modified the circuit card locations and thereby diversified the power supplies so that the trip units have less dependency on common fuses. Additional corrective actions include performing an engineering evaluation to determine if there are additional single component failures, operator errors, or events for the FW heating system that could result in a drop in FW temperature of greater than 100°F.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Clinton Power Station, Unit 1	05000461	2017	- 007	- 02

NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

General Electric -- Boiling Water Reactor, 3473 Megawatts Thermal Rated Core Power
Energy Industry Identification System (EIIS) codes are identified in text as [XX].

EVENT IDENTIFICATION

Manual Reactor SCRAM due to Loss of Feedwater Heating

A. Plant Operating Conditions before the Event

Unit: 1	Event Date: 6/10/17	Event Time: 2256 CDT
Mode: 1	Mode Name: Power Operation	Reactor Power: 98 percent

B. Description of Event

On June 10, 2017, at 2256 CDT, Clinton Power Station (CPS) experienced a complete loss of the ‘A’ feedwater (FW) heater [HX] string. The operators received numerous FW trouble alarms on FW string 'A' and low pressure (LP) heater [HTR] 1A/1B bypass valve (1CB004) automatically opened. The operators entered procedure CPS 4005.01, "Loss of FW Heating," which directs the operators to restore and maintain reactor power at or below the original power level and within stability control and power/flow map limits by adjusting reactor recirculation flow, control rods, or CRAM array. The operators lowered core flow and inserted all CRAM rods. The operators observed that FW temperature had dropped by greater than 100°F. Procedure CPS 4005.01 directs the operators to place the reactor mode switch [JS] into the shutdown position and enter procedure CPS 4100.01, "Reactor Scram." With the unit at approximately 93 percent power, the operators placed the mode switch in shutdown at 2306 on June 10, 2017 and entered procedure CPS 4100.01.

The components of the CPS power conversion system are designed to produce electrical power utilizing the steam generated by the reactor [RCT], condense that steam into water, and return the water to the reactor as heated feedwater. A portion of the main turbine [TRB] steam is extracted for FW heating. CPS has two trains of cascading FW heaters. Under normal, full power conditions, the extraction steam valves [V] to each of the FW heaters are open such that steam is condensed in the body of the heater. In addition, the normal heater drain valves are normally open and the emergency FW heater drain valves to the main condenser [COND] are closed. A high-high level in a FW heater will isolate the input sources to the heater (i.e., extraction steam valve(s) and the upstream normal FW heater drain valve(s)), reducing the reactor FW temperature.

A walkdown of panel [PL] 1PA08J (Miscellaneous Sensors & Transducers Power Supply Cabinet), which houses Moore trip units for both the ‘A’ and ‘B’ FW heating strings, identified that there were no lights on rack CA-1 and the indicator for fuse [FU] FU-89 was open.

NARRATIVE

Troubleshooting using an ohmmeter found that high resistance in the circuit was eliminated after pulling Moore trip unit 1LYHD103A, which is the trip unit that provides automatic actions on a Hi-Hi level for FW heater 4A. This indicated that the loss of power to rack CA-1 was caused by fuse FU-89 opening in response to a shorted condition on the Moore trip unit 1LYHD103A. When fuse FU-89 opened, power was also lost to the Moore trip units for FW heaters 1A, 2A, 3A, 5A, and 6A resulting in the loss of heating to the 'A' FW heater string.

C. Cause of the Event

The root cause of the manual reactor scram due to the loss of the 'A' feedwater heater string is that the design of the feedwater heater level control trip circuitry was not adequate to prevent scrams due to an unevaluated single point vulnerability. The first contributing cause is a technical error in an analysis that incorrectly determined that there was no single component failure that will cause a FW temperature drop greater than 100°F. The second contributing cause is that the designer did not adequately consider the potential for high heat conditions inside panel 1PA08J due to lack of adequate cooling; the high heat conditions in the panel have resulted in shortened life and reduced reliability of the Moore trip units.

D. Safety Consequences

The event which caused the unplanned reactor scram did not involve any personnel or nuclear safety consequences. It is reportable under the provisions of 10 CFR 50.73(a)(2)(iv)(A) due to the manual actuation of the reactor protection system. This event is also reportable under 10 CFR 50.73(a)(2)(ii)(B) as it is also considered an unanalyzed condition.

An assessment of the safety consequences and implication of this event determined that the manual reactor scram ensured the plant remained in a safe and stable condition and no operating limits were exceeded.

The design basis loss of feedwater heating transient for CPS is based on a maximum temperature transient of 100°F. Should this event occur at a lower reactor power level, the severity of the transient would be reduced commensurate with the reduction in FW heating.

The purpose of the 100°F limit for the feedwater temperature reduction is to ensure that, combined with a turbine trip and bypass failure, no fuel cladding damage or fuel rod perforations are expected to occur and the peak bottom vessel pressure remains well below the ASME Level B Service limit. The data from this loss of feedwater heating event was reviewed by the Exelon Nuclear Fuels Safety Analysis Group. Although the change in feedwater temperature exceeded the 100°F assumed in the loss of feedwater heating analysis, it was concluded that sufficient margin existed to safety limits.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Clinton Power Station, Unit 1	05000461	2017	- 007	- 02

NARRATIVE

E. Corrective Actions

Prior to startup, CPS modified the circuit card locations in the panel that contains the Heater Drain system Moore trip units and thereby diversified the power supplied so that the trip units have less dependency on common fuses. In addition, the blown fuse FU-89 was replaced. Additional corrective actions included installation of temporary cooling and temperature loggers in the Panel 1PA08J to monitor for elevated temperature condition.

Further actions include developing an engineering evaluation to determine if there are additional single component failures, operator errors, or events for the FW heating system that could result in a decrease in FW temperature of greater than 100°F. In addition, a permanent modification that eliminates the high heat conditions in the panel is being tracked by the corrective action program.

F. Previous Similar Occurrences

LER 88-025 - Loss of Feedwater Heating System Transient Outside Design Basis Due to Inadequate Communication Between the Architect Engineer and the Nuclear Steam Supply System Supplier.

On July 28, 1988, CPS experienced a partial loss of FW heating. The FW temperature drop, excluding the change caused by a reduction in power, was greater than 102°F, but less than 112°F. The design basis loss of FW heating transient for CPS is based on a maximum temperature transient of 100°F. The cause of the loss of FW heating was the inappropriate setting of the FW heater level controllers. The cause of exceeding the design basis is attributed to the failure of the FW heating system design to meet design requirements. This was caused by a lack of adequate communication between the Nuclear Steam Supply System (NSSS) supplier and the architect engineer regarding the NSSS design requirements for the FW heating system. Feedwater heating system design changes, including changes to the level trip setpoint for closing the extraction steam valves and replacing power supply fuses, were made to ensure that the design basis is met.

G. Component Failure Data

Failed card was determined to be a Moore Industries DCA alarm card.

Model Number: DCA/4-20ma/DH1L2/45dC/-AD-100HB1 (PC)
Serial Number: 2412651